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Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)

Appendices

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Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)

Appendices

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Abstract

During plant operation, the walls of reactor pressure vessels (RPVs) are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the core area. If an embrittled RPV had a flaw of critical size and certain severe system transients were to occur, the flaw could very rapidly propagate through the vessel, resulting in a through-wall crack and challenging the integrity of the RPV. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by a rapid cooling of the internal RPV surface in combination with repressurization of the RPV. Advancements in our understanding and knowledge of materials behavior, our ability to realistically model plant systems and operational characteristics, and our ability to better evaluate PTS transients to estimate loads on vessel walls led the NRC to realize that the earlier analysis, conducted in the course of developing the PTS Rule in the 1980s, contained significant conservatisms.

This report summarizes 21 supporting documents that describe the procedures used and results obtained in the probabilistic risk assessment, thermal hydraulic, and probabilistic fracture mechanics studies conducted in support of this investigation. Recommendations on toughness-based screening criteria for PTS are provided.

Foreword

The reactor pressure vessel is exposed to neutron radiation during normal operation. Over time, the vessel steel becomes progressively more brittle in the region adjacent to the core. If a vessel had a preexisting flaw of critical size and certain severe system transients occurred, this flaw could propagate rapidly through the vessel, resulting in a through-wall crack. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by rapid cooling (i.e., thermal shock) of the internal reactor pressure vessel surface that may be combined with repressurization. The simultaneous occurrence of critical-size flaws, embrittled vessel, and a severe PTS transient is a very low probability event. The current study shows that U.S. pressurized-water reactors do not approach the levels of embrittlement to make them susceptible to PTS failure, even during extended operation well beyond the original 40-year design life.

Advancements in our understanding and knowledge of materials behavior, our ability to realistically model plant systems and operational characteristics, and our ability to better evaluate PTS transients to estimate loads on vessel walls have shown that earlier analyses, performed some 20 years ago as part of the development of the PTS rule, were overly conservative, based on the tools available at the time. Consistent with the NRC's Strategic Plan to use best-estimate analyses combined with uncertainty assessments to resolve safety-related issues, the NRC's Office of Nuclear Regulatory Research undertook a project in 1999 to develop a technical basis to support a risk-informed revision of the existing PTS Rule, set forth in Title 10, Section 50.61, of the Code of Federal Regulations (10 CFR 50.61).

Two central features of the current research approach were a focus on the use of realistic input values and models and an explicit treatment of uncertainties (using currently available uncertainty analysis tools and techniques). This approach improved significantly upon that employed in the past to establish the existing 10 CFR 50.61 embrittlement limits. The previous approach included unquantified conservatisms in many aspects of the analysis, and uncertainties were treated implicitly by incorporating them into the models.

This report summarizes a series of 21 reports that provide the technical basis that the staff will consider in a potential revision of 10 CFR 50.61; it includes a description of analysis procedures and a detailed discussion of findings. The risk from PTS was determined from the integrated results of the Fifth Version of the Reactor Excursion Leak Analysis Program (RELAP5) thermal-hydraulic analyses, fracture mechanics analyses, and probabilistic risk assessment. These calculations demonstrate that, even through the period of license extension, the likelihood of vessel failure attributable to PTS is extremely low ($\approx 10^{-8}$/year) for all domestic pressurized water reactors. Limited analyses are continuing to further evaluate this finding. Should the $\approx 10^{-8}$/year value be confirmed, this would provide a basis for significant relaxation, or perhaps elimination, of the embrittlement limit established in 10 CFR 50.61. Such changes would reduce unnecessary conservatism without affecting safety because the operating reactor fleet has little probability of exceeding the limits on the frequency of reactor vessel failure established from NRC guidelines on core damage frequency and large early release frequency through the period of license extension.

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Executive Summary

This report summarizes the results of a 5-year study conducted by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES). The aim of this study was to develop the technical basis for revision of the Pressurized Thermal Shock (PTS) Rule, as set forth in Title 10, Section 50.61, of the *Code of Federal Regulations* (10 CFR 50.61), "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," consistent with the NRC's current guidelines on risk-informed regulation. This report, together with other supporting reports documenting the study details and results, provides this basis.

This executive summary begins with a description of PTS, how it might occur, and its potential consequences for the reactor pressure vessel (RPV). This is followed by a summary of the current regulatory approach to PTS, which leads directly to a discussion of the motivations for conducting this project. Following this introductory information, we describe the approach used to conduct the study, and summarize our key findings and recommendations, which include a proposal for revision of the PTS screening limits. We then conclude the executive summary with a discussion of the potential impact of this proposal on regulations other than 10 CFR 50.61.

Description of PTS

During the operation of a nuclear power plant, the RPV walls are exposed to neutron radiation, resulting in localized embrittlement of the vessel steel and weld materials in the area of the reactor core. If an embrittled RPV had an existing flaw of critical size and certain severe system transients were to occur, the flaw could propagate very rapidly through the vessel, resulting in a through-wall crack and challenging the integrity of the RPV. The severe transients of concern, known as pressurized thermal shock (PTS), are characterized by a rapid cooling (i.e., thermal shock) of the internal RPV surface and downcomer, which may be followed by repressurization of the RPV. Thus, a PTS event poses a potentially significant challenge to the structural integrity of the RPV in a pressurized-water reactor (PWR).

A number of abnormal events and postulated accidents have the potential to thermally shock the vessel (either with or without significant internal pressure). These events include a pipe break or stuck-open valve in the primary pressure circuit, a break of the main steam line, etc. During such events, the water level in the core drops as a result of the contraction produced by rapid depressurization. In events involving a break in the primary pressure circuit, an additional drop in water level occurs as a result of leakage from the break. Automatic systems and operators must provide makeup water in the primary system to prevent overheating of the fuel in the core. However, the makeup water is much colder than that held in the primary system. As a result, the temperature drop produced by rapid depressurization coupled with the near-ambient temperature of the makeup water produces significant thermal stresses in the thick section steel wall of the RPV. For embrittled RPVs, these stresses could be sufficient to initiate a running crack, which could propagate all the way through the vessel wall. Such through-wall cracking of the RPV could precipitate core damage or, in rare cases, a large early release of radioactive material to the environment. Fortunately, the coincident occurrence of critical-size flaws, embrittled vessel steel and weld material, and a severe PTS transient is a very low-probability event. In fact, only a few currently operating PWRs are projected to closely approach the current statutory limit on the level of embrittlement during their planned operational life.

Current Regulatory Approach to PTS

As set forth in 10 CFR 50.61, the PTS Rule requires licensees to monitor the embrittlement of their RPVs using a reactor vessel material surveillance program qualified under Appendix H to 10 CFR Part 50, "Reactor Vessel Material Survellience Program Requirements." The surveillance results are then used together with the formulae and tables in 10 CFR 50.61 to estimate the fracture toughness transition temperature (RT_{NDT}) of the steels in the vessel's beltline and how those transition temperatures increase as a result of irradiation damage throughout the operational life of the vessel. For licensing purposes, 10 CFR 50.61 provides instructions on how to use these estimates of the effect of irradiation damage to estimate the value of RT_{NDT} that will occur at end of license (EOL), a value called RT_{PTS}. 10 CFR 50.61 also provides "screening limits" (maximum values of RT_{NDT} permitted during the plant's operational life) of +270°F (132°C) for axial welds, plates, and forgings, and +300°F (149°C) for circumferential welds. These screening limits correspond to a limit of $5\text{x}10^{-6}$ events/year on the annual probability of developing a through-wall crack [RG 1.154]. Should RT_{PTS} exceed these screening limits, 10 CFR 50.61 requires the licensee to either take actions to keep RT_{PTS} below the screening limit (by implementing "reasonably practicable" flux reductions to reduce the embrittlement rate, or by deembrittling the vessel by annealing [RG 1.162]), or perform plant-specific analyses to demonstrate that operating the plant beyond the 10 CFR 50.61 screening limit does not pose an undue risk to the public [RG 1.154].

While no currently operating PWR has an RT_{PTS} value that exceeds the 10 CFR 50.61screening limit before EOL, several plants are close to the limit (3 are within 2°F, while 10 are within 20°F). Those plants are likely to exceed the screening limit during the 20-year license renewal period that is currently being sought by many operators. Moreover, some plants maintain their RT_{PTS} values below the 10 CFR 50.61 screening limits by implementing flux reductions (low-leakage cores, ultra-low-leakage cores), which are fuel management strategies that can be economically deleterious in a deregulated marketplace. Thus, the 10 CFR 50.61 screening limits can restrict both the licensable and economic lifetime of PWRs.

Motivation for this Project

It is now widely recognized that the state of knowledge and data limitations in the early 1980s necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis for the current PTS Rule. The most prominent of these conservatisms include the following factors:

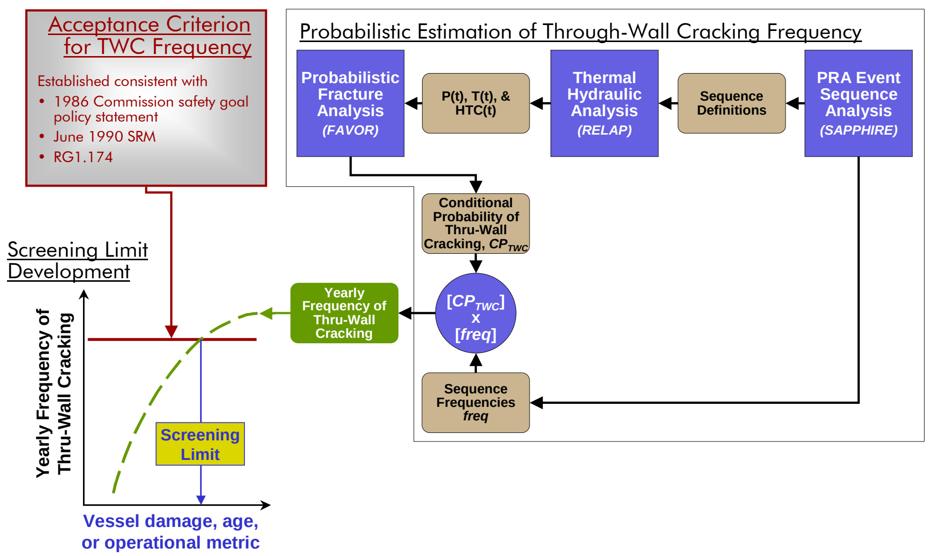
- highly simplified treatment of plant transients (very coarse grouping of many operational sequences (on the order of 10^5) into very few groups ($\approx$10), necessitated by limitations in the computational resources needed to perform multiple thermal-hydraulic calculations)
- lack of any significant credit for operator action
- characterization of fracture toughness using RT_{NDT}, which has an intentional conservative bias
- use of a flaw distribution that places *all* flaws on the interior surface of the RPV, and, in general, contains larger flaws than those usually detected in service
- a modeling approach that treated the RPV as if it were made entirely from the most brittle of its constituent materials (welds, plates, or forgings)
- a modeling approach that assessed RPV embrittlement using the peak fluence over the entire interior surface of the RPV

These factors indicate the high likelihood that the current 10 CFR 50.61 PTS screening limits are unnecessarily conservative. Consequently, the NRC staff believed that reexamining the technical basis for these screening limits, based on a modern understanding of all the factors that influence PTS, would most likely provide strong justification for substantially relaxing these limits. For these reasons, RES undertook this study with the objective of developing the technical basis to support a risk-informed revision of the PTS Rule and the associated PTS screening limit.

Approach

As illustrated in the following figure, three main models (shown as solid blue squares), taken together, allow us to estimate the annual frequency of through-wall cracking in an RPV:

- probabilistic risk assessment (PRA) event sequence analysis
- thermal-hydraulic (TH) analysis
- probabilistic fracture mechanics (PFM) analysis



Schematic showing how a probabilistic estimate of through-wall cracking frequency (TWCF) is combined with a TWCF acceptance criterion to arrive at a proposed revision of the PTS screening limit

First, a PRA event sequence analysis is performed to define the sequences of events that are likely to cause a PTS challenge to RPV integrity, and estimate the frequency with which such sequences can be expected to occur. The event sequence definitions are then passed to a TH model that estimates the temporal variation of temperature, pressure, and heat-transfer coefficient in the RPV downcomer, which is characteristic of each sequence definition. These temperature, pressure, and heat-transfer coefficient histories are then passed to a PFM model that uses the TH output, along with other information concerning plant design and construction materials, to estimate the time-dependent "driving force to fracture" produced by a particular event sequence. The PFM model then compares this estimate of fracture driving force to the fracture toughness, or fracture resistance, of the RPV steel. This comparison allows us to estimate the probability that a crack could grow to sufficient size that it would penetrate all the way through the RPV wall if that particular sequence of events actually occured. The final step in the analysis involves a simple matrix multiplication of the probability of through-wall cracking (from the PFM analysis) with the frequency at which a particular event sequence is expected to occur (as defined by the event-tree analysis). This product establishes an estimate of the annual frequency of through-wall cracking that can be expected for a particular plant after a particular period of operation when subjected to a particular sequence of events. The annual frequency of through-wall cracking is then summed for all event sequences to estimate the total annual frequency of through-wall cracking for the vessel. Performance of such analyses for various operating lifetimes provides an estimate of how the annual frequency of through-wall cracking can be expected to vary over the lifetime of the plant.

The probabilistic calculations just described are performed to establish the technical basis for a revised PTS Rule within an integrated systems analysis framework. Our approach considers a broad range of factors that influence the likelihood of vessel failure during a PTS event, while accounting for uncertainties in these factors across a breadth of technical disciplines. Two central features of this approach are a focus on the use of realistic input values and models (wherever possible), and an *explicit* treatment of uncertainties (using currently available uncertainty analysis tools and techniques). Thus, our current approach improves upon that employed in developing SECY-82-465, which included intentional and unquantified conservatisms in many aspects of the analysis, and treated uncertainties *implicitly* by incorporating them into the models.

Key Findings

The findings from this study are divided into the following five topical areas: (1) the expected magnitude of the through-wall cracking frequency (TWCF) for currently anticipated operational lifetimes, (2) the material factors that dominate PTS risk, (3) the transient classes that dominate PTS risk, (4) the applicability of these findings (based on detailed analyses of three PWRs) to PWRs *in general*, and (5) the annual limit on TWCF established consistent with current guidelines on risk-informed regulation. In this summary, ***the conclusions are presented in boldface italic***, while the supporting information is shown in regular type.

TWCF Magnitude for Currently Anticipated Operational Lifetimes

- ***The degree of PTS challenge is low for currently anticipated lifetimes and operating conditions.***
  - Even at the end of license extension (60 operational years, or 48 effective full-power years (EFPY) at an 80% capacity factor), the mean estimated TWCF does not exceed 2×10^{-8}/year for the plants analyzed. Considering that the RPVs at the Beaver Valley Power Station and Palisades Nuclear Power Plant are constructed from some of the most irradiation-sensitive materials in commercial reactor service today, these results suggest that, provided that operating practices do not change dramatically in the future, the operating reactor fleet is in little danger of exceeding either the TWCF limit of 5×10^{-6}/yr expressed by Regulatory Guide 1.154 [RG 1.154] or the value of 1×10^{-6}/yr recommended in Chapter 10 of this report — even after license extension.

Material Factors and their Contributions to PTS Risk

- ***Axial flaws, and the toughness properties that can be associated with such flaws, control nearly all of the TWCF.***
  - Axial flaws are much more likely than circumferential flaws to propagate through the RPV wall because the applied fracture driving force increases continuously with increasing crack depth for an axial flaw. Conversely, circumferentially oriented flaws experience a driving force peak mid-wall, providing a natural crack arrest mechanism. It should be noted that crack initiation from circumferentially oriented flaws is likely; it is only their through-wall propagation that is much less likely (relative to axially oriented flaws).
  - It is, therefore, the toughness properties that can be associated with axial flaws that control nearly all of the TWCF. These include the toughness properties of plates and axial welds at the flaw locations. Conversely, the toughness properties of both circumferential welds and forgings have little effect on the TWCF because these can be associated only with circumferentially oriented flaws.

Transients and their Contributions to PTS Risk

- ***Transients involving primary side faults are the dominant contributors to TWCF, while transients involving secondary side faults play a much smaller role.***
  - The severity of a transient is controlled by a combination of three factors:
    - initial cooling rate, which controls the thermal stress in the RPV wall
    - minimum temperature of the transient, which controls the resistance of the vessel to fracture
    - pressure retained in the primary system, which controls the pressure stress in the RPV wall
  - The significance of a transient (i.e., how much it contributes to PTS risk) depends on these three factors and the likelihood that the transient will occur.
  - Our analysis considered transients in the following classes (as shown in the following table):
    - primary side pipe breaks
    - stuck-open valves on the primary side
    - main steam line breaks
    - stuck-open valves on the secondary side
    - feed-and-bleed
    - steam generator tube rupture
    - mixed primary and secondary initiators

Factors contributing to the severity and risk-dominance of various transient classes

Transient Class		Transient Severity			Transient Likelihood	TWCF Contribution
		Cooling Rate	Minimum Temperature	Pressure		
Primary Side Pipe Breaks	Large-Diameter	Fast	Low	Low	Low	Large
	Medium-Diameter	Moderate	Low	Low	Moderate	Large
	Small-Diameter	Slow	High	Moderate	High	~0
Stuck-Open Valves, Primary Side	Valve Recloses	Slow	Moderate	High	High	Large
	Valve Remains Open	Slow	Moderate	Low	High	~0
Main Steam Line Break		Fast	Moderate	High	High	Small
Stuck-Open Valve(s), Secondary Side		Moderate	High	High	High	~0
Feed-and- Bleed		Slow	Low	Low	Low	~0
Steam Generator Tube Rupture		Slow	High	Moderate	Low	~0
Mixed Primary & Secondary Initiators		Slow	Mixed		Very Low	~0

Color Key	Enhances TWCF Contribution	Intermediate	Diminishes TWCF Contribution

  - The table above provides a qualitative summary our results for these transient classes in terms of both transient severity and the likelihood that the transient will occur. The color-coding of table entries indicates the contribution (or lack thereof) of these factors to the TWCF of the various classes of transients. This summary indicates that the risk-dominant transients (medium- and large-diameter primary side pipe breaks, and stuck-open primary side valves that later reclose) all have multiple factors that, in combination, result in their significant contributions to TWCF.
    - For medium- to large-diameter primary side pipe breaks, the fast to moderate cooling rates and low downcomer temperatures (generated by rapid depressurization and emergency injection of low-temperature makeup water directly to the primary) combine to produce a high-severity transient. Despite the moderate to low likelihood that these transients will occur, their severity (if they do occur) makes them significant contributors to the total TWCF.

- For stuck-open primary side valves that later reclose, the repressurization associated with valve reclosure coupled with low temperatures in the primary combine to produce a high-severity transient. This, coupled with a high likelihood of transient occurrence, makes stuck-open primary side valves that later reclose significant contributors to the total TWCF.
- The small or negligible contribution of all secondary side transients (main steam line break, stuck-open secondary valves) results directly from the lack of low temperatures in the primary system. For these transients, the minimum temperature of the primary for times of relevance is controlled by the boiling point of water in the secondary (212°F (100°C) or above). At these temperatures, the fracture toughness of the RPV steel is sufficiently high to resist vessel failure in most cases.

Applicability of These Findings to PWRs in General

- ***Credits for operator action, while included in our analysis, do not influence these findings in any significant way.*** Operator action credits can dramatically influence the risk-significance of ***individual*** transients. Therefore, appropriate credits for operator action need to be included as part of a "best estimate" analysis because there is no way to establish *a priori* if a particular transient will make a large contribution to the total risk. Nonetheless, the results of our analyses demonstrate that these operator action credits have a small ***overall effect*** on a plant's ***total TWCF***, for reasons detailed below.
    - <u>Medium- and Large-Diameter Primary Side Pipe Breaks</u>: No operator actions are modeled for any break diameter because, for these events, the safety injection systems do not fully refill the upper regions of the reactor coolant system (RCS). Consequently, operators would never take action to shut off the pumps.
    - <u>Stuck-Open Primary Side Valves that May Later Reclose</u>: Reasonable and appropriate credit for operator actions (throttling of the high-pressure injection (HPI) system) has been included in the PRA model. However, these credits have a small influence on the estimated values of vessel failure probability attributable to transients caused by a stuck-open valve in the primary pressure circuit (SO-1 transients) because the credited operator actions only prevent repressurization when SO-1 transients initiate from Hot Zero Power (HZP) conditions and when the operators act promptly (within 1 minute) to throttle the HPI. Complete removal of operator action credits from the model only slightly increases the total risk associated with SO-1 transients.
    - <u>Main Steam Line Breaks</u>: For the overwhelming majority of transients caused by a main steam line break (MSLB), vessel failure is predicted to occur between 10 and 15 minutes after transient initiation because the thermal stresses associated with the rapid cooldown reach their maximum within this timeframe. Thus, all of the long-term effects (isolation of feedwater flow, timing of HPSI control) that can be influenced by operator actions have no effect on vessel failure probability because such factors influence the progression of the transient *after failure has occurred* (if it occurs at all). Only factors affecting the initial cooling rate (i.e., plant power level at time of transient initiation, break location inside or outside of containment) can influence the conditional probability of through-wall cracking (CPTWC), and operator actions do not influence such factors in any way.
- ***Because the severity of the most significant transients in the dominant transient classes is controlled by factors that are common to PWRs in general, the TWCF results presented herein can be used with confidence to develop revised PTS screening criteria that apply to the entire fleet of operating PWRs.***
    - <u>Medium- and Large-Diameter Primary Side Pipe Breaks</u>: For these break diameters, the fluid in the primary cools faster than the wall of the RPV. In this situation, ***only*** the thermal conductivity of the steel and the thickness of the RPV wall control the thermal stresses and, thus, the severity of the fracture challenge. Perturbations in the fluid cooldown rate controlled by break diameter, break location, and season of the year do not play a role. Thermal conductivity is a physical property,

so it is very consistent for all RPV steels, and the thicknesses of the three RPVs analyzed are typical of PWRs. Consequently, the TWCF contribution of medium- to large-diameter primary side pipe breaks is expected to be consistent from plant-to-plant and can be well represented for all PWRs by the analyses reported herein.

  - Stuck-Open Primary Side Valves that May Later Reclose: A major contributor to the risk-significance of SO-1 transients is the return to full system pressure once the valve recloses. The operating and safety relief valve pressures of all PWRs are similar. Additionally, as previously noted, operator action credits only slightly affect the total risk associated with this transient class.
  - Main Steam Line Breaks: Since MSLBs fail early (within 10–15 minutes after transient initiation), only factors affecting the initial cooling rate can have any influence on the CPTWC values. These factors, which include the plant power level at event initiation and the location of the break (inside or outside of containment), are not influenced by operator actions in any way.

- ***Sensitivity studies performed on the TH and PFM models to investigate the effect of credible model variations on the predicted TWCF values revealed no effects significant enough to recommend changes to the baseline RELAP and FAVOR models, or to recommend cautions regarding the robustness of those models.***
- ***An investigation of design and operational characteristics for five additional PWRs revealed no differences in sequence progression, sequence frequency, or plant thermal-hydraulic response significant enough to call into question the applicability of the TWCF results from the three detailed plant analyses to PWRs in general.***
- ***An investigation of potential external initiating events (e.g., fires, earthquakes, floods) revealed that the contribution of those events to the total TWCF can be regarded as negligible.***

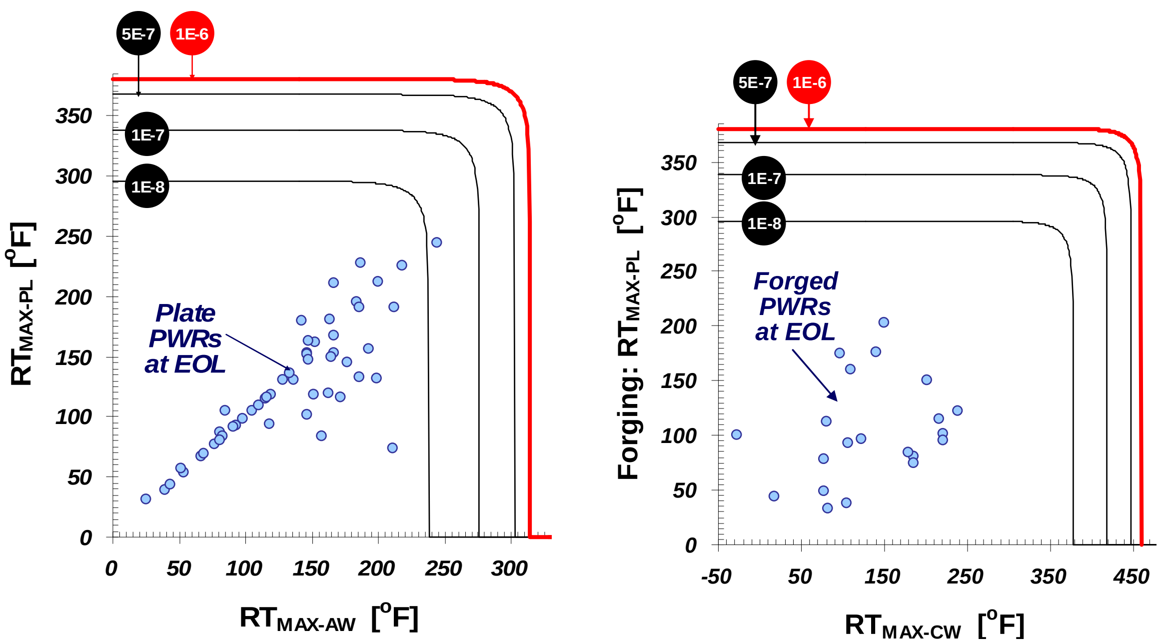
Annual Limit on TWCF

- ***The current guidance provided by Regulatory Guide 1.174 [RG 1.174] for large early release is appropriately applied to setting an acceptable annual TWCF limit of $1x10^{-6}$ events/year.***
  - While many post-PTS accident progressions led only to core damage (which suggests a TWCF limit of $1x10^{-5}$ events/year limit in accordance with Regulatory Guide 1.174), uncertainties in the accident progression analysis led to our recommendation to adopt the more conservative limit of $1x10^{-6}$ events/year based on LERF.

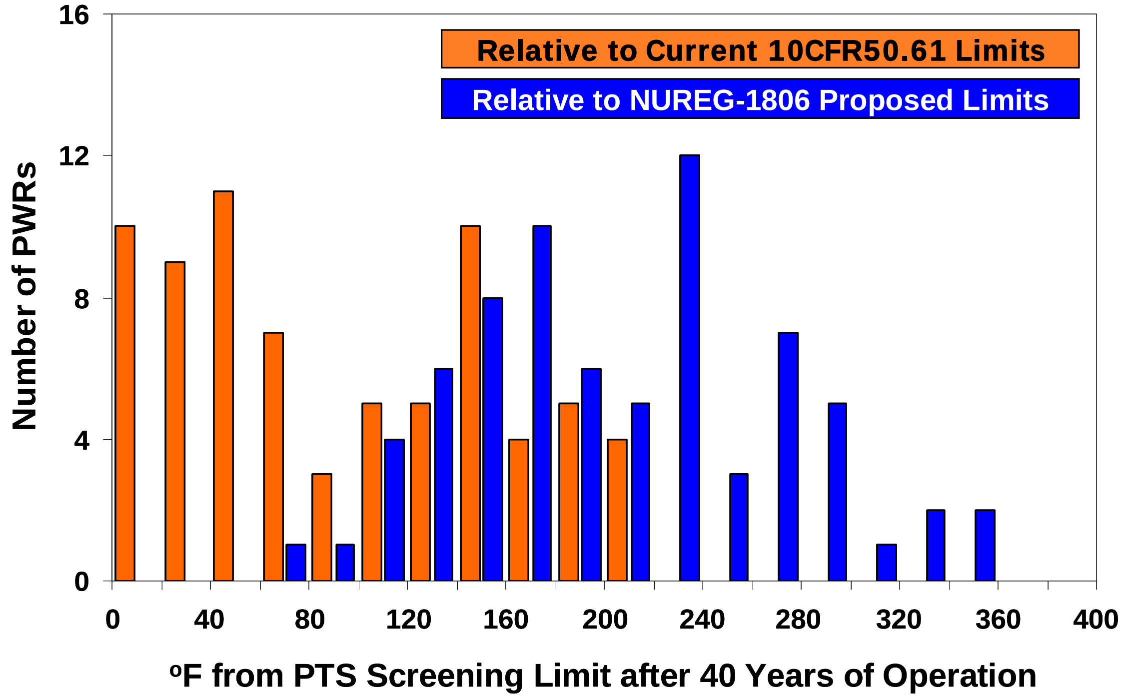
Recommended Revision of the PTS Screening Limits

We recommend using different reference temperature (*RT*) metrics to characterize an RPV's resistance to fractures initiating from different flaws at different locations in the vessel. Specifically, we recommend a reference temperature for flaws occurring along axial weld fusion lines (RT_{AW} or RT_{AW-MAX}), another for flaws occurring in plates or in forgings (RT_{PL} or RT_{PL-MAX}), and a third for flaws occurring along circumferential weld fusion lines (RT_{CW} or RT_{CW-MAX}). In each of these reference temperature pairs, the first metric is a weighted value that accounts for the differences between plants in weld fusion line area or plate volume, while the second metric is a maximum value that can be estimated based only on the information in the NRC's Reactor Vessel Integrity Database (RVID). We also recommend using different *RT* values together to characterize the fracture resistance of the vessel's beltline region, in recognition of the fact that the probability of vessel fracture initiating from different flaw populations varies considerably in response to factors that are both understood and predictable. Correlations between these *RT* metrics and the TWCF attributable to axial weld flaws, plate flaws, and circumferential weld flaws show little plant-to-plant variability because of the general similarity of PTS challenges among plants.

RT-based screening limits were established by setting the total TWCF (i.e., that attributable to axial weld flaws and plate flaws and circumferential weld flaws) equal to the reactor vessel failure frequency acceptance criterion of $1x10^{-6}$ events per year. The following figures graphically represent these screening limits (for the maximum *RT* metrics), along with an assessment of all operating PWRs relative to these limits. In these figures, the region of the graphs between the red locus and the origin has TWCF values below the $1x10^{-6}$ acceptance criterion, so these combinations of reference temperatures would be considered acceptable and require no further analysis. By contrast, the region of the graph outside of the red locus has TWCF values above the $1x10^{-6}$ acceptance criterion, indicating the need for additional analysis or other measures to justify continued plant operation. Clearly, operating PWRs do not closely approach the $1x10^{-6}$/year limit. At EOL, at least 70°F, and up to 290°F, (39 to 161°C) separate plate-welded PWRs from the proposed screening limit; this separation between plant-specific values and the proposed screening limit reduces by 10–20°F (5.5 to 11°C) at end of license extension (EOLE, defined as 60 operating years or 48 EFPY). Additionally, no forged plant is anywhere close to the limit of $1x10^{-6}$ events per year at either EOL or EOLE. This separation of operating plants from the screening limit contrasts markedly with the current situation, where the most embrittled plants are within 1°F (0.5°C) of the screening limit set forth in 10 CFR 50.61. These differences in the "proximity" of operating plants to the current (10 CFR 50.61) and proposed screening limits are illustrated by the bar graph on the next page.



Comparison of RT-based screening limits (curves) with assessment points for operating PWRs at EOL (Left: plate vessels, Right: forged vessels)



Difference between the proximity of operating PWRs to the current RT_{PTS} screening limits and to the screening limits proposed based on the work presented in this report.

These *RT*-based screening limits (and similar limits described in the text for application to weighted *RT* values) apply to PWRs in general, subject only to the following provisos:

- When assessing a forged vessel where the forging has a very high reference temperature (RT_{PL} above 225°F (107°C)) ***and*** the forging is believed to be susceptible to subclad cracking, a plant-specific analysis of the TWCF produced by the subclad cracks should be performed. However, no forging is projected to reach this level of embrittlement, even at EOLE.
- When assessing an RPV having a wall thickness of 7-in. (18-cm) or less (7 vessels), the proposed *RT* limits are conservative.
- When assessing an RPV having a wall thickness of 11-in. (28-cm) or greater, the proposed *RT* limits may be nonconservative. For the three plants meeting this criterion, either the *RT* limits would need to be reduced or known conservatisms in the current analysis would have to be removed to demonstrate compliance with the TWCF limit of 1×10^{-6} event/year. However, because these three plants are Units 1, 2, and 3 of the Palo Verde Nuclear Generating Station, which have vessels with very low embrittlement projected at EOL and EOLE, there is little practical need for such plant-specific analysis.

Aside from relying on different *RT* metrics than 10 CFR 50.61, this proposed revision of the PTS screening limit differs from the current screening limit in the absence of a "margin term." Use of a margin term is appropriate to account (at least approximately) for factors that occur in application but were not considered in the analysis upon which the screening limit is based. For example, the 10 CFR 50.61 margin term accounts for uncertainty in copper, nickel, and initial RT_{NDT}. However, our model explicitly considers uncertainty in all of these variables, and represents these uncertainties as being larger (a conservative representation) than would be appropriate in any plant-specific application of the proposed screening limit. Consequently, use of the 10 CFR 50.61 margin term with the new screening limits is inappropriate. In general, the following additional reasons suggest that use of *any* margin term with the proposed screening limits is inappropriate:

(1) The *TWCF* values used to establish the screening limit represent 90th percentile values or greater.

(2) The results from our three plant-specific analyses apply to PWRs *in general*, as demonstrated in Chapters 8 and 9 of this report.

(3) Certain aspects of our modeling cannot reasonably be represented as “best estimates.” On balance, there is a conservative bias to these non-best-estimate aspects of our analysis because residual conservatisms in the model far outweigh residual nonconservatisms.

Abbreviations

¼-T FLAW	Surface-breaking flaw defined by ASME Boiler and Pressure Vessel Code as having a depth equal to one-quarter of the vessel wall thickness and a length equal to six times the flaw depth
1D	One-Dimensional
ABAQUS	Commercial finite element code developed by Hibbett, Karlsson, and Sorenson in Pawtucket, Rhode Island
ACRS	Advisory Committee on Reactor Safety (NRC)
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
APET	Accident Progression Event Tree
APEX	Advanced Plant Experiment
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient without Scram
B&W	Babcock and Wilcox
BWOG	Babcock and Wilcox Owners' Group
BCC	Body-Centered Cubic
BWR	Boiling-Water Reactor
CDF	Core Damage Frequency
CE	Combustion Engineering
CEOG	Combustion Engineering Owners' Group
CFD	Computational Fluid Dynamics
CL	Cold Leg
CFR	*Code of Federal Regulations*
CFT	Core Flood Tank
CPI	Conditional Probability of Crack Initiation
CPTWC	Conditional Probability of Through-Wall Cracking
CSAU	Code Scaling, Applicability, and Uncertainty Methodology
CSNI	Committee on the Safety of Nuclear Installations
CST	Condensate Storage Tank
CVN	Charpy V-Notch
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EFPY	Effective Full-Power Years
EFW	Emergency Feedwater
EOL	End of License (40 operating years, 32 EFPY)
EOLE	End of License Extension (60 operating years, 48 EFPY)

EPRI	Electric Power Research Institute
ESFAS	Engineered Safety Features Actuation System
F&B	Feed-and-Bleed
FAVOR	Fracture Analysis of Vessels, Oak Ridge
FCI	Frequency of Crack Initiation
GMAW	Gas Metal Arc Weld
H2TS	Hierarchical, Two-Tiered Scaling
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HPI	High-Pressure Injection
HPSI	High-Pressure Safety Injection
HRA	Human Reliability Analysis
HSSI	Heavy Section Steel Irradiation (Project)
HZP	Hot Zero Power
IAEA	International Atomic Energy Agency
ID	Inner Diameter
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPTS	Integrated Pressurized Thermal Shock
ISLOCA	Interfacing Systems Loss-of-Coolant Accident
ITV	Intermediate Test Vessel
IVO	Imatran Voima Oy
LAS	Low-Alloy Steel
LBLOCA	Large-Break Loss-of-Coolant Accident (pipe diameters above ~8-in. (~20-cm))
LEFM	Linear Elastic Fracture Mechanics
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LOF	Lack of Inter-Run Fusion
LOFT	Loss-of-Fluid Test facility
LPI	Low-Pressure Injection
LPSI	Low-Pressure Safety Injection
MBLOCA	Medium-Break Loss-of-Coolant Accident (pipe diameters of ~4 to 8-in. (~10 to 20-cm))
MFIV	Main Feedwater Isolation Valve
MFW	Main Feedwater
MIST	Multi-loop Integral System Test
MRJ	Materials Reliability Project
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break

NDT	Nil-Ductility Temperature
NEA	Nuclear Energy Agency (OECD)
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NUREG/CR	NRC Technical Report Designator (Contractor-prepared Report published by the U.S. Nuclear Regulatory Commission)
OD	Outer Diameter
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PFM	Probabilistic Fracture Mechanics
PIRT	Phenomena Identification and Ranking Table
PNNL	Pacific Northwest National Laboratories
PORV	Power-Operated Relief Valve
Ppb	Parts per Billion
PRA	Probabilistic Risk Assessment
PRODIGAL	Probability of Defect Initiation and Growth Analysis
PTS	Pressurized Thermal Shock
PTSE	Pressurized Thermal Shock Experiment
PVRUF	Pressure Vessel Research Users' Facility
PWR	Pressurized-Water Reactor
QHO	Quantitative Health Objective, as defined by the Commission's Safety Goal Policy Statement [NRC FR 86]
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RELAP	Reactor Leak and Power excursion code
REMIX	a computer program used to determine the temperature of a plume in the downcomer when the flow in the loops is stagnant
RES	Office of Nuclear Regulatory Research (NRC)
RG	Regulatory Guide
RLE	Review-Level Earthquake
ROSA	Rig of Safety Assessment
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RT	Reference Temperature
RVFF	Reactor Vessel Failure Frequency
RVID	Reactor Vessel Integrity Database
RWST	Refueling Water Storage Tank
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SAW	Submerged Arc Weld
SBLOCA	Small-Break Loss-of-Coolant Accident (pipe diameters below ~4-in. (~10-cm))
SCC	Stress Corrosion Cracking
SECY	Secretary of the (U.S. Nuclear Regulatory) Commission

SEMISCALE	a 1:1705 scaled experimental facility that simulates the primary system of a 4-loop PWR plant
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SMAW	Submerged Metal Arc Weld
SO-1	Stuck-open valve in the primary pressure circuit
SO-2	Stuck-open valve in the secondary pressure circuit
SQA	Software Quality Assurance
SRM	Staff Requirements Memorandum
SRV	Safety/Relief Valve
SSC	System, Structure, or Component
SSE	Safe-Shutdown Earthquake
SSRV	Secondary System Relief Valve
TBV	Turbine Bypass Valve
TH	Thermal-Hydraulics
TMI	Three Mile Island
TSE	Thermal Shock Experiment
TWCF	Through-Wall Cracking Frequency
UMD	University of Maryland
UPTF	Upper Plenum Test Facility
USE	Charpy V-Notch Upper-Shelf Energy
V&V	Verification and Validation
VCIF	Vessel Crack Initiation Frequency
(W)	Westinghouse
WOG	Westinghouse Owners' Group
WPS	Warm Pre-Stress

Nomenclature

Symbols Used in Thermal-Hydraulics

α	thermal diffusivity, m^2/s
β	bulk coefficient of expansion, 1/C
μ	viscosity, kg/m-s
ν	kinematic viscosity, m^2/s
ρ	density, kg/m^3
σ	stress, kg/s^2
τ	characteristic time
C_p	heat capacity, m^2/s^2-C
g	gravitational acceleration, m/s^2
Gr	Grashof Number
h	convective heat transfer coefficient, W/m^2-C
D	diameter, m
J	joules, kg-m^2/s^2
k	conductivity, W/m-C
l	length, m
Nu	Nusselt Number
Pr	Prandtl Number
P	pressure, kg/m-s^2
q	heat flux, W/m^2
Re	Reynolds Number
Ri	Richardson Number
s	seconds
t	thickness, m
t	time, s
u	velocity, m/s
T	temperature, C
W	watts, kg-m^2/s^3

Symbols Used in Fracture Mechanics

$2a$	Flaw depth measured through the vessel wall thickness
$2c$	Flaw length measured parallel to the axial or circumferential direction of the vessel
Cu	Copper content, weight%
J_{Ic}	A fracture toughness measure defined by ASTM E1820, which quantifies the resistance of metals to crack initiation by the initiation, growth, and coalescence of microvoids
J-R	A fracture toughness measure defined by ASTM E1820, which quantifies the resistance of metals to ductile tearing
K_{Jc}	A fracture toughness measure defined by ASTM E1921, which quantifies the resistance of metals to crack initiation by cleavage mechanisms
K_{Ia}	A fracture toughness measure defined by ASTM E1221, which quantifies the ability of metals to arrest (stop) a running cleavage crack
K_{Ic}	A fracture toughness measure defined by ASTM E399, which quantifies the resistance of metals to crack initiation under plane strain conditions
$K_{Ic(min)}$	The minimum K_{Ic} fracture toughness possible at a particular temperature
$K_{APPLIED}$	Linear elastic crack driving force
$\mathcal{L}$	For a buried defect, distance from the wetted clad surface on the vessel ID to the inner crack tip
ℓ	The length of the fusion line of an axial weld
Ni	Nickel content, weight%
P	Phosphorus content, weight%
RT_{AW}	A fracture toughness reference temperature, which characterizes the RPV's resistance to fractures initiating from flaws found along the axial weld fusion lines. It corresponds to the maximum RT_{NDT} of the plates/welds that lie to either side of the weld fusion lines, and is weighted to account for differences in weld fusion line length (and, therefore, number of simulated flaws) between vessel courses.
RT_{PL}	A fracture toughness reference temperature, which characterizes the RPV's resistance to fractures initiating from flaws found in plates that are not associated with welds. It corresponds to the maximum RT_{NDT} occurring anywhere in the plate.
RT_{CW}	A fracture toughness reference temperature, which characterizes the RPV's resistance to fractures initiating from flaws found along the circumferential weld fusion lines. It corresponds to the maximum RT_{NDT} of the plates/welds that lie to either side of the weld fusion lines.
RT_{NDT}	Transition fracture toughness reference temperature defined by ASME NB-2331
$RT_{NDT(u)}$	Unirradiated value of RT_{NDT}
RT_{PTS}	RT_{NDT} projected end of license to account for the effects of irradiation (defined in 10 CFR 50.61)
t_{WALL}	Vessel wall thickness
t_{CLAD}	Stainless steel cladding thickness

T_{30}	The temperature at which the mean CVN energy is 30 ft-lbs (41J)
$T_{35/50}$	Charpy V-notch energy transition temperature defined as the temperature at which the CVN energy is at least 50 ft-lbs (68J) and the lateral expansion of the specimen is at least 0.035-in. (0.89-mm) [See the definition on page 2-7]
T_{NDT}	Nil-ductility temperature defined by ASTM E-208
ΔT_{30}	The shift in the CVN 30 ft-lb (41J) transition temperature produced by radiation damage
σ_{flow}	Flow strength, average of tensile yield and tensile ultimate strength
ϕt	Fluence

Glossary

Terms Used in Probabilistic Risk Assessment

Abnormal operating procedure	A procedure (i.e., list of actions) used to address unique or special plant circumstances identified while using emergency operating procedures (EOPs). These abnormal operating procedures are usually called by EOPs, but may be indicated directly by some plant conditions.
Accident progression event tree	The event tree used to model the part of the accident sequence that follows the onset of core damage, including containment response to severe accident conditions, equipment availability, and operator performance.
Binning	The process of taking a large number of sequences and combining then into a smaller number of groups, that are expected to have similar characteristics (e.g., TH conditions), to allow effective utilization of limited resources.
Core damage	Uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release.
Dominant scenario	An accident sequence (scenario) that is usually represented by the top 10 or 20 events or groups of events modeled in a PRA, which accounts for a large fraction of the specified end state.
Emergency operating procedure	The primary procedure (i.e., list of actions) used to respond to a plant disturbance resulting from an initiating event.
Event tree	A logic diagram that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.
Fault tree	A deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events.
Large Early Release	The rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions, such that there is a potential for early health effects.
Latin Hypercube sampling	A stratified sampling technique, in which the random variable distributions are divided into equal probability intervals, and probabilities are then randomly selected from within each interval.
Mitigating equipment	Systems or components, used to respond to an initiating event, of which successful operation prevents the occurrence of an undesired event or state.
Pre-initiator human failure event	Human failure events that represent the impact of human errors committed during actions performed prior to the initiation of an accident (e.g., during maintenance or the use of calibration procedures).
Post-initiator human failure event	Human failure events that represent the impact of human errors committed during actions performed in response to an accident initiator.

Prompt fatality	A fatality that results from substantial radiation exposures incurred during short time periods (usually within weeks, though up to 1 year for pulmonary effects).
PTS bin	A group of sequences that are expected to have similar TH characteristics and are represented by one unique set of TH characteristics during a FAVOR calculation.
Risk-informed	An approach to analyzing and evaluating activities, which bases decisions on the results of traditional engineering evaluations, supported by insights derived from the use of PRA methods.
Scenario	See Sequence.
Screening	The process of eliminating items from further consideration based on their negligible contribution to the probability of an undesired end state or its consequences.
Sequence	A representation in terms of an initiating event followed by a sequence of failures or successes of events (i.e., system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., potential for PTS).

Terms Used in Thermal-Hydraulics

Blowdown	Rapid depressurization of a system in response to a break.
Break flow	Flow of water (liquid and vapor) out a pipe break or a valve.
Break energy	Energy content of the fluid flow out a break.
Bottom-up	To break up a complex system into its subsystems, and then break up each subsystem into its components, examine individual local phenomena and processes that most affect each component, and build up the total complex system from these individual pieces (like manufacturing a car).
Coast down	Time required for a pump to stop rotating once power is shut off due to inertia.
Decay heat	Heat generated from radioactive decay of fission products.
Enthalpy	Sum of internal energy and volume multiplied by pressure.
Flash	Change of phase from saturated liquid to vapor resulting from decrease in pressure.
Flow quality	Mass fraction of flow stream that is steam. Higher quality flow would have a high mass fraction of steam.
Forced flow	Flow driven by a pump.
Inventory	Mass of water.
Loop flow	Mass flow rate of coolant in a circuit.
Makeup water	Water reservoir available for inventory control.
Natural circulation	Flow driven by buoyancy (gravity).
Pressure drop	Change in pressure due to conversion of mechanical energy to internal energy.
Protection system	Electrical controls to actuate engineering safety features.
Quality	Mass fraction of steam in a two-phase steam-water mixture.

Saturation temperature	A temperature corresponding to phase change from liquid to vapor.
Sensible heat	The product of specific heat and temperature change of subcooled liquid.
Subcooled	A system is *subcooled* if it exists entirely in a liquid state. The *degree of subcooling* is the number of degrees that the temperature of the system would have to be raised to cause boiling.
Throttled	Operation of a control valve to regulate flow.
Top-down	To characterize a complex system by establishing the governing behavior, or phenomenon, that is most important, and then proceed from that starting point to successive lower levels, by identifying the processes that have the greatest influence on the top-level phenomenon.
Trip	A "trip" occurs when a breaker opens in response to its trip mechanism (an arm that holds the breaker closed moves to allow the breaker to open). When a reactor trips, all of the breakers that provide power to the rod control system open, causing the rods to be inserted in the core and stopping the nuclear reaction. When a pump trips, the breaker opens, thereby disconnecting power and causing the pump to stop.
Water solid	A situation in which there is no steam in the system (i.e., it is all liquid). A "water solid" system is *subcooled*.

Terms Used in Fracture Mechanics

Brittle	Fracture occurring without noticeable macroscopic plastic deformation (stretching) of the material.
Cleavage fracture	Microscopically, cleavage is a fracture mode that occurs preferentially along certain atomic planes through the grains of the material. Cleavage can only occur in ferritic steels (i.e., steels having a body-centered cubic lattice structure). Macroscopically, cleavage fracture is often called "brittle" fracture because little noticeable plastic deformation (stretching) of the material occurs. (Note, however, that plastic flow at the micro-scale is a necessary precursor to cleavage.) Macroscopically, cleavage fracture is also characterized as being a sudden event, with cracks of very large dimensions developing over durations measured in fractional seconds. A useful, although inexact, analogue for cleavage fracture in common experience is the breaking of glass.
Ductile fracture	Microscopically, ductile fracture occurs through the initiation, growth, and eventual coalescence of micro-voids in the material into a macroscopic crack. These micro-voids tend to initiate at local heterogeneities in the material (e.g., inclusions, carbides, clusters of dislocations). Macroscopically, ductile fracture is associated with considerable plastic deformation (stretching) of the material. Relative to cleavage fracture, ductile fracture occurs very slowly, with crack growth rates measured in seconds rather than in micro-seconds (for cleavage).
Fracture toughness	A general term referring to a material's resistance to fracture. The term may be modified to refer to fractures by different mechanisms: Arrest fracture toughness measures a material's ability to stop a running cleavage crack. Cleavage fracture toughness measures a material's ability to resist crack initiation in cleavage. Ductile fracture toughness measures a material's ability to resist crack initiation attributable to ductile mechanisms on the upper shelf.

Lower shelf	At low temperatures, the toughness behavior of steels occurs by transgranular cleavage and is said to be on the lower shelf. On the lower shelf, a fracture is unstable, and is often referred to as a "brittle" fracture.
Reference temperature	A characteristic temperature used to locate the transition curve of a ferritic steel on the temperature axis.
Transition (or transition curve)	Between lower shelf and upper shelf temperatures, the fracture behavior of a ferritic material is said to be in "transition." At low temperatures in transition, fracture occurs by cleavage. As temperature increases through the transition regime, fracture occurs by ductile crack initiation and growth, a process which is terminated by cleavage. At still higher temperatures, cleavage cannot occur, and upper shelf conditions exist.
Upper shelf	At high temperatures, the toughness behavior of steels occurs by ductile mechanisms (micro-void initiation, growth, and coalescence) and is said to be on the upper shelf. On the upper shelf, afracture is stable and dissipates considerable amounts of energy.

Terms Used in Uncertainty Analysis

Aleatory	Aleatory uncertainties arise as a result of the randomness inherent in a physical or human process. Consequently, aleatory uncertainties are fundamentally irreducible. If the uncertainty in a variable is characterized as being aleatory, the entire distribution of the variable is carried through each simulation run.
Epistemic	Epistemic uncertainties are caused by limitations in our current state of knowledge (or understanding) of a given process. Epistemic uncertainties can, in principle, be reduced by an increased state of knowledge. If the uncertainty in a variable is characterized as being epistemic in a probabilistic simulation, individual values of the variable are randomly selected from a distribution and propagated through the calculation. This procedure models the understanding that the "correct" value of the variable is knowable, at least in principal. Thus, for epistemic uncertainties, individual simulation runs are deterministic, while the totality of all simulation runs captures the uncertainty characteristic of the epistemic variable.

Appendix A – Master Transient List and FAVOR 04.1 Results Summary

Table A.1. Transient descriptions and FAVOR 04.1 results for medium- and large-diameter pipe break (LOCA) transients

#	Dia [in]	System Failure	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
				32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																			
9	16	2.54-cm [16-in.] hot leg break	6.99E-06	5.0	5.8	6.5	7.6	1.5	3.3	10.5	10.2	6.40E-04	3.02E-03	1.21E-02	7.44E-02	8.63E-07	1.43E-05	1.58E-04	2.66E-03
7	8	2.54-cm [8-in.] surge line break	2.11E-05	14.4	15.7	17.2	17.4	17.2	28.6	19.4	14.9	6.16E-04	2.58E-03	9.06E-03	4.66E-02	2.01E-06	2.08E-05	1.44E-04	1.52E-03
117	5.7	14.366-cm [5.657-in.] cold leg break, summer conditions (HHSI, LHSI temp = 55°F, Accumulator Temp = 105°F)	2.11E-05	0.2	0.2	0.2	0.2	0.0	0.0	0.0	0.0	1.27E-05	4.66E-05	1.36E-04	6.45E-04	8.29E-10	2.29E-08	9.46E-08	1.14E-06
116	5.7	14.366-cm [5.657-in.] cold leg break with break area increased 30%	1.81E-05	0.0	0.0	0.1	0.1	0.0	0.0	0.0	0.0	1.27E-06	9.46E-06	4.85E-05	4.33E-04	2.15E-10	1.27E-08	6.23E-08	2.16E-06
56	4	10.16-cm [4.0-in.] surge line break (see Note at end of table)	1.23E-04	79.4	77.1	74.2	68.9	13.2	30.8	51.4	43.2	7.43E-04	2.84E-03	9.36E-03	4.46E-02	8.76E-07	1.07E-05	8.29E-05	9.82E-04

#	Dia [in]	System Failure	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
				32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Oconee Unit 1																			
156	16	40.64-cm [16-in.] hot leg break. ECC suction switch to the containment sump included in the analysis.	7.03E-06	6.8	13.2	26.9	28.5	0.0	0.1	4.6	6.3	1.63E-06	1.74E-05	2.81E-03	1.55E-02	2.88E-10	6.82E-09	6.21E-06	7.83E-05
164	8	20.32-cm [8 in.] surge line break. ECC suction switch to the containment sump included in the analysis.	2.12E-05	41.2	38.9	36.1	39.7	0.1	1.2	17.1	29.6	1.20E-06	1.09E-05	1.29E-03	7.06E-03	7.78E-11	9.05E-09	6.85E-06	8.05E-05
160	5.7	14.37-cm [5.656-in.] surge line break. ECC suction switch to the containment sump included in the analysis.	1.82E-05	33.3	39.8	32.4	26.3	0.1	0.6	24.6	32.8	2.77E-06	1.97E-05	1.32E-03	5.84E-03	1.36E-09	4.01E-08	1.33E-05	1.21E-04
172	4	10.16-cm [4-in.] cold leg break. ECC suction switch to the containment sump included in the analysis.	1.06E-04	0.0	0.1	0.8	1.5	-	-	0.5	1.2	4.90E-10	8.71E-09	6.86E-06	6.29E-05	-	-	9.55E-08	2.27E-06
Palisades																			
40	16	40.64-cm (16-in.) hot leg break. Containment sump recirculation included in the analysis.	3.22E-05	65.3	63.7	58.2	58.7	14.4	20.5	27.7	30.5	7.60E-04	1.76E-03	1.03E-02	6.42E-02	1.36E-05	7.11E-05	1.21E-03	9.57E-03
62	8	20.32-cm (8-in.) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	7.07E-06	6.9	6.6	6.6	6.9	2.1	3.2	4.2	4.7	3.73E-04	8.85E-04	5.31E-03	3.37E-02	1.08E-05	5.50E-05	8.57E-04	6.59E-03
63	5.7	14.37-cm (5.656-in.) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	6.06E-06	2.0	2.0	2.3	2.3	0.8	1.4	2.0	2.1	1.41E-04	3.37E-04	2.08E-03	1.30E-02	5.00E-06	2.72E-05	4.53E-04	3.30E-03

#	Dia [in]	System Failure	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
				32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
58	4	10.16-cm (4-in.) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	2.66E-04	11.5	11.7	11.8	11.3	4.9	8.8	12.6	12.4	2.01E-05	4.79E-05	3.05E-04	1.78E-03	8.11E-07	4.47E-06	7.88E-05	5.34E-04
64	4	10.16-cm (4-in.) surge line break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	7.07E-06	4.0	3.8	3.4	3.3	2.2	3.5	3.7	3.4	2.35E-04	5.28E-04	2.95E-03	1.68E-02	1.21E-05	6.03E-05	7.92E-04	4.88E-03
59	4	10.16-cm (4-in.) cold leg break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	2.09E-04	0.6	0.8	1.0	1.2	0.1	0.3	0.7	1.0	1.34E-06	3.94E-06	3.32E-05	2.15E-04	2.00E-08	1.72E-07	5.73E-06	5.37E-05

Note: There are no operator actions for any of these transients, and all transients initiate from full power conditions **except for** Beaver Valley 56, which initiates from hot zero power conditions. However, Beaver Valley 56 is used to represent full power conditions in this analysis.

Table A.2. Transient descriptions and FAVOR 04.1 results for small-diameter pipe break (LOCA) transients

#	Dia [in]	System Failure	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
				32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																			
114	2.8	7.18-cm [2.828-in.] surge line break, summer conditions (HHSI, LHSI temp = 55°F, Accumulator Temp = 105°F), heat transfer coefficient increased 30% (modeled by increasing heat transfer surface area by 30% in passive heat structures).	9.76E-05	0.24	0.35	0.54	0.70	0.00	0.01	0.03	0.08	2.55E-06	1.52E-05	6.70E-05	4.57E-04	6.63E-11	8.69E-09	7.25E-08	2.45E-06
115	2.8	7.18-cm [2.828-in.] cold leg break	9.76E-05	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
3	2	5.08-cm [2-in.] surge line break	9.76E-05	0.00	0.02	0.03	0.08	-	0.00	0.00	0.01	3.81E-08	6.85E-07	4.01E-06	5.79E-05	-	2.28E-10	2.18E-09	3.57E-07
2	1.4	3.59-cm [1.414-in.] surge line break	1.23E-04	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Oconee Unit 1																			
154	3.4	8.53-cm [3.36-in.] surge line break [Break flow area reduced by 30% from 10.16-cm [4-in.] break]. Vent valves do not function. ECC suction switch to the containment sump included in the analysis.	1.34E-04	0.00	0.00	0.15	0.36	-	-	0.10	0.79	0.00E+00	1.56E-10	1.18E-06	1.44E-05	-	-	1.44E-08	1.08E-06
178	3.4	8.53-cm [3.36-in.] surge line break [Break flow area reduced by 30% from 10.16-cm [4-in.] break]. Vent valves do not function. ECC suction switch to the containment sump included in the analysis.	2.12E-05	0.00	0.00	0.04	0.07	-	-	0.03	0.21	0.00E+00	1.56E-10	1.18E-06	1.44E-05	-	-	1.44E-08	1.08E-06
141	3.2	8.19-cm [3.22-in.] surge line break [Break flow area increased by 30% from 7.18-cm [2.828-in.] break].	1.06E-04	0.64	1.40	1.71	1.45	0.00	0.06	1.92	2.66	2.00E-08	2.05E-07	1.54E-05	6.44E-05	2.40E-13	7.57E-10	3.77E-07	4.21E-06

#	Dia [in]	System Failure	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
				32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
142	2.4	6.01-cm [2.37-in.] surge line break [Break flow area decreased by 30% from 7.18-cm [2.828-in.] break].	1.06E-04	-	-	-	0.00	-	-	-	-	-	-	-	2.24E-09	-	-	-	-
145	1.7	4.34-cm [1.71-in.] surge line break [Break flow area increased by 30% from 3.81-cm [1.5-in.] break]. Winter conditions assumed [HPI, LPI temp = 277 K [40° F] and CFT temp = 294 K [70° F]].	1.34E-04	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Palisades																			
61	2.8	7.18-cm (2.8-in.) cold leg break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	2.09E-04	0.01	0.03	0.11	0.16	0.00	0.01	0.04	0.11	2.98E-08	1.89E-07	3.87E-06	3.06E-05	3.57E-10	3.32E-09	3.71E-07	6.55E-06
60	2	5.08-cm (2-in.) surge line break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	2.09E-04	1.67	1.94	2.60	2.56	0.89	1.69	3.00	3.51	3.82E-06	1.07E-05	8.22E-05	5.02E-04	1.81E-07	1.13E-06	2.41E-05	1.88E-04
2	1.4	3.59-cm (1.414-in.) surge line break. Containment sump recirculation included in the analysis.	2.66E-04	-	-	-	0.00	-	-	-	0.00	-	-	-	1.41E-14	-	-	-	2.82E-15

<u>Note</u>: There are no operator actions for any of these transients, and all transients initiate from full power conditions.

Table A.3. Transient descriptions and FAVOR 04.1 results for stuck-open primary valve transients (including value reclosure)

TH #	Transients Including Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																				
126	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 6,000 s	Operator controls HHSI 10 minutes after allowed.	N	1.87E-04	0.2	0.1	0.0	0.0	23.6	14.0	3.1	0.77	8.24E-07	2.57E-06	2.79E-06	1.40E-05	8.19E-07	2.50E-06	2.50E-06	1.04E-05
60	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 6,000 s.	None.	N	2.15E-05	0.1	0.1	0.0	0.0	16.8	7.5	1.8	0.41	4.69E-06	1.17E-05	1.39E-05	4.24E-05	4.69E-06	1.17E-05	1.39E-05	4.20E-05
130	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 3,000 s at HZP	Operator controls HHSI 10 minutes after allowed.	Y	3.09E-05	0.1	0.1	0.0	0.0	16.5	7.1	1.0	0.24	4.14E-06	8.46E-06	6.53E-06	3.69E-05	4.14E-06	8.45E-06	6.37E-06	1.99E-05
97	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 3,000 s.	None.	Y	3.74E-06	0.0	0.0	0.0	0.0	4.0	2.0	0.6	0.13	7.68E-06	1.82E-05	1.77E-05	6.03E-05	7.68E-06	1.82E-05	1.77E-05	5.27E-05
129	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 6,000 s at HZP	Operator controls HHSI 10 minutes after allowed.	Y	3.09E-05	0.0	0.0	0.0	0.0	2.9	1.2	0.2	0.03	9.54E-07	1.62E-06	1.76E-06	2.08E-05	9.54E-07	1.61E-06	1.60E-06	1.99E-05
123	Reactor/turbine trip w/two stuck-open pressurizer SRVs, which reclose at 3,000 s at HZP	Operator controls HHSI 10 minutes after allowed.	Y	1.65E-07	0.0	0.0	0.0	0.0	1.2	0.6	0.3	0.09	6.32E-05	1.78E-04	3.90E-04	1.81E-03	5.59E-05	1.50E-04	2.86E-04	1.38E-03
71	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 6,000 s.	None.	Y	3.74E-06	0.0	0.0	0.0	0.0	1.2	0.5	0.1	0.02	1.96E-06	4.00E-06	3.57E-06	1.94E-05	1.96E-06	4.00E-06	3.49E-06	8.10E-06

TH #	Transients Including Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
61	Reactor/turbine trip w/two stuck-open pressurizer SRV, which recloses at 3,000 s.	None.	N	1.79E-06	0.0	0.0	0.0	0.0	0.2	0.3	0.4	0.26	5.67E-06	2.93E-05	1.15E-04	7.15E-04	1.20E-06	8.87E-06	4.14E-05	4.14E-04
69	Reactor/turbine trip w/two stuck-open pressurizer SRVs, which reclose at 3,000 s.	None.	Y	2.09E-08	0.0	0.0	0.0	0.0	0.3	0.2	0.1	0.04	1.32E-04	4.34E-04	1.18E-03	5.68E-03	9.65E-05	3.09E-04	8.31E-04	4.72E-03
120	Reactor/turbine trip w/two stuck-open pressurizer SRVs, which recloses at 6,000 s	Operator controls HHSI 10 minutes after allowed.	N	9.98E-07	0.0	0.0	0.0	0.0	0.5	0.2	0.0	0.01	3.62E-06	1.14E-05	3.17E-05	2.63E-04	3.07E-06	6.71E-06	6.64E-06	2.69E-05
124	Reactor/turbine trip w/two stuck-open pressurizer SRVs, which reclose at 6,000 s at HZP	Operator controls HHSI 10 minutes after allowed.	Y	1.65E-07	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.01	1.39E-05	5.49E-05	2.00E-04	1.27E-03	4.16E-06	8.91E-06	1.17E-05	1.06E-04
92	Reactor/turbine trip w/two stuck-open pressurizer SRVs, one recloses at 3000 s.	None.	Y	2.13E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.01	5.15E-05	2.35E-04	8.71E-04	4.82E-03	2.04E-07	1.87E-06	1.15E-05	1.21E-04
93	Reactor/turbine trip w/two stuck-open pressurizer SRVs. One valve recloses at 6000 seconds, while the other valve remains open.	None.	Y	2.13E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.01	5.15E-05	2.35E-04	8.71E-04	4.82E-03	2.04E-07	1.87E-06	1.15E-05	1.21E-04
70	Reactor/turbine trip w/two stuck-open pressurizer SRVs, which reclose at 6,000 s.	None.	Y	2.09E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	1.35E-05	5.60E-05	2.32E-04	1.82E-03	6.03E-06	1.25E-05	1.78E-05	1.25E-04
66	Reactor/turbine trip w/two stuck-open pressurizer SRVs. One valve recloses at 3000 seconds, while the other valve remains open.	None.	N	1.18E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.01	5.25E-06	2.80E-05	1.13E-04	7.05E-04	1.42E-08	2.15E-07	1.41E-06	1.83E-05
62	Reactor/turbine trip w/two stuck-open pressurizer SRV, which recloses at 6,000 s.	None.	N	1.08E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	1.20E-06	4.64E-06	1.61E-05	1.57E-04	9.61E-07	2.12E-06	2.14E-06	7.76E-06

TH #	Transients Including Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
59	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 3,000 s.	None.	N	3.46E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	0.00E+00	4.10E-12	1.97E-07	0.00E+00	0.00E+00	0.00E+00	1.61E-08
67	Reactor/turbine trip w/two stuck-open pressurizer SRVs. One valve recloses at 6000 seconds, while the other valve remains open.	None.	N	1.18E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	2.07E-07	2.21E-06	1.24E-05	1.38E-04	0.00E+00	2.51E-10	4.49E-09	5.39E-07
119	Reactor/turbine trip w/two stuck-open pressurizer SRV, which recloses at 6,000 s	Operator controls HHSI 1 minute after allowed.	N	6.84E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	5.60E-07	4.78E-06	2.57E-05	2.51E-04	1.76E-12	5.42E-09	6.34E-08	2.81E-06
121	Reactor/turbine trip w/two stuck-open pressurizer SRV, which recloses at 3,000 s at HZP	Operator controls HHSI 1 minute after allowed	Y	1.33E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	9.79E-06	4.63E-05	1.91E-04	1.26E-03	6.92E-11	1.90E-08	5.74E-07	2.89E-05
122	Reactor/turbine trip w/two stuck-open pressurizer SRVs, which reclose at 6,000 s at HZP	Operator controls HHSI 1 minute after allowed.	Y	1.33E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	9.79E-06	4.63E-05	1.91E-04	1.26E-03	3.57E-12	7.81E-09	1.14E-07	5.81E-06
125	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 6,000 s	Operator controls HHSI 1 minute after allowed.	N	1.34E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	2.39E-09	7.20E-08	3.27E-07	4.87E-06	0.00E+00	5.58E-19	7.77E-11	1.16E-08
127	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 6,000 s at HZP	Operator controls HHSI 1 minute after allowed.	Y	2.59E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	4.37E-18	8.95E-09	1.62E-07	1.75E-05	0.00E+00	0.00E+00	0.00E+00	1.18E-07
128	Reactor/turbine trip w/one stuck-open pressurizer SRV, which recloses at 3,000 s at HZP	Operator controls HHSI 1 minute after allowed.	Y	2.59E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	4.37E-18	8.95E-09	1.62E-07	1.75E-05	0.00E+00	0.00E+00	0.00E+00	1.18E-07

Oconee Unit 1

TH #	Transients Including Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
122	Stuck-open pressurizer safety valve. Valve recloses at 6000 secs.	Operator throttles HPI at 10 minutes after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Y	7.57E-06	13.7	4.7	0.4	0.3	76.8	74.5	26.9	11.27	3.01E-06	7.23E-06	4.68E-05	1.44E-04	3.01E-06	7.23E-06	4.68E-05	1.44E-04
165	Stuck-open pressurizer safety valve. Valve recloses at 6000 secs [RCS low pressure point].	None	Y	1.76E-06	4.0	1.3	0.1	0.1	22.4	20.3	7.8	2.26	2.75E-06	6.55E-06	4.24E-05	1.24E-04	2.75E-06	6.55E-06	4.24E-05	1.24E-04
124	Stuck-open pressurizer safety valve. Valve recloses at 3000 secs.	Operator throttles HPI at 10 minutes after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Y	7.57E-06	0.1	0.2	0.2	0.2	0.5	3.0	12.8	9.02	3.61E-08	2.81E-07	1.48E-05	9.38E-05	3.60E-08	2.80E-07	1.48E-05	9.37E-05
168	TT/RT with stuck-open pzr SRV. SRV assumed to reclose at 3000 secs.	None	Y	1.76E-06	0.0	0.0	0.0	0.0	0.1	0.4	2.4	1.33	5.48E-08	3.75E-07	1.78E-05	1.10E-04	5.45E-08	3.73E-07	1.78E-05	1.09E-04
113	Stuck-open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 10 minutes after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling)	N	5.07E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.30	0.00E+00	0.00E+00	0.00E+00	1.42E-07	0.00E+00	0.00E+00	0.00E+00	1.31E-07
109	Stuck-open pressurizer safety valve. Valve recloses at 6000 secs [RCS low pressure point].	None	N	9.58E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	0.00E+00	1.31E-09	1.83E-07	0.00E+00	0.00E+00	1.30E-09	1.83E-07
112	Stuck-open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 1 minute after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 27 K [50°F] subcooling)	N	1.25E-04	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
114	Stuck-open pressurizer safety valve. Valve recloses at 3000 secs.	After valve recloses, operator throttles HPI 1 minute after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling)	N	1.25E-04	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

TH #	Transients Including Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
115	Stuck-open pressurizer Safety Valve. Valve recloses at 3000 secs.	After valve recloses, operator throttles HPI 10 minutes after 2.7 K [5°F] subcooling and 254-cm [100”] pressurizer level is reached (throttling criteria is 50°F subcooling)	N	5.07E-05	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
121	Stuck-open pressurizer safety valve. Valve recloses at 6000 secs .	Operator throttles HPI at 1 minute after 2.7 K [5°F] subcooling and 254-cm [100”] pressurizer level is reached [throttling criteria is 27.8 K [50°F] subcooling].	Y	2.28E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	0.00E+00	6.54E-11	2.06E-07	0.00E+00	0.00E+00	0.00E+00	1.28E-08
123	Stuck-open pressurizer safety valve. Valve recloses at 3000 secs.	Operator throttles HPI at 1 minute after 2.7 K [5°F] subcooling and 254-cm [100”] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Y	2.28E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	0.00E+00	6.54E-11	2.06E-07	0.00E+00	0.00E+00	0.00E+00	1.28E-08
149	TT/RT with stuck-open pzr SRV. SRV assumed to reclose at 3000 secs.	None	N	9.58E-06	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Palisades																				
65	One stuck-open pressurizer SRV that recloses at 6000 sec after initiation. Containment spray is assumed not to actuate.	None	Y	1.24E-04	6.5	5.8	4.4	2.4	67.2	45.4	17.5	8.40	2.60E-05	5.50E-05	2.57E-04	8.40E-04	2.53E-05	5.40E-05	2.55E-04	8.37E-04
48	Two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None	Y	7.67E-07	0.1	0.1	0.1	0.0	1.4	0.9	0.3	0.12	8.57E-05	1.67E-04	6.50E-04	1.96E-03	8.46E-05	1.66E-04	6.47E-04	1.95E-03
53	Turbine/reactor trip with two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None	N	1.09E-03	0.0	0.0	0.3	0.4	0.0	0.2	0.8	1.27	9.91E-10	3.34E-08	1.48E-06	1.23E-05	3.86E-10	1.62E-08	1.13E-06	1.15E-05

TH #	Transients Including Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
42	Turbine/reactor trip with two stuck-open pressurizer SRVs. Containment spray is assumed not to actuate.	Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60%.	N	7.67E-07	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table A.4. Transient descriptions and FAVOR 04.1 results for stuck-open primary valve transients (no value reclosure)

TH #	Transients Without Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																				
14	Reactor/turbine trip w/one stuck-open pressurizer SRV	None.	N	2.23E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	1.56E-11	2.02E-08	9.33E-08	2.93E-06	0.00E+00	0.00E+00	1.09E-15	3.80E-10
34	Reactor/turbine trip w/two stuck-open pressurizer SRVs	None.	N	4.95E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	2.39E-07	2.53E-06	1.41E-05	1.53E-04	1.60E-17	1.64E-09	1.56E-08	1.03E-06
64	Reactor/turbine trip w/two stuck-open pressurizer SRVs	None.	Y	8.67E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	7.51E-06	4.39E-05	2.18E-04	1.80E-03	1.28E-09	7.65E-08	8.92E-07	2.43E-05
94	Reactor/turbine trip w/one stuck-open pressurizer SRV.	None.	Y	4.10E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	4.70E-11	2.14E-08	7.76E-06	0.00E+00	0.00E+00	0.00E+00	4.73E-08
Oconee Unit 1																				
169	TT/RT with stuck-open pzr SRV [valve flow area reduced by 30%]. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]]. Vent valves do not function.	None	Y	7.33E-06	0.0	0.0	0.1	0.2	0.0	0.0	0.6	0.95	1.90E-10	2.65E-08	1.47E-05	9.92E-05	0.00E+00	3.12E-11	7.13E-07	1.00E-05
146	TT/RT with stuck-open pzr SRV [valve flow area reduced by 30%]. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]]. Vent valves do not function.	None	N	4.23E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.17	0.00E+00	0.00E+00	9.28E-08	3.91E-06	0.00E+00	0.00E+00	7.44E-09	8.90E-07
147	TT/RT with stuck-open pzr SRV. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]].	None	N	3.63E-05	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

TH #	Transients Without Valve Reclosure System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
148	TT/RT with partially stuck-open pzr SRV [flow area equivalent to 1.5 in diameter opening]. HTC coefficients increased by 1.3.	None	N	4.23E-05	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
170	TT/RT with stuck-open pzr SRV. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]].	None	Y	6.28E-06	-	-	0.0	0.0	-	-	-	0.00	-	-	6.68E-12	1.38E-07	-	-	-	7.72E-09
171	TT/RT with partially stuck-open pzr SRV [flow area equivalent to 1.5 in diameter opening]. HTC coefficients increased by 1.3.	None	Y	7.33E-06	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table A.5. Transient descriptions and FAVOR 04.1 results for large-diameter steam line break transients

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																				
103	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Y	1.1E-05	0.07	0.17	0.30	0.63	0.74	2.54	4.78	5.17	7.36E-06	6.67E-05	4.14E-04	4.68E-03	3.96E-07	8.57E-06	7.41E-05	1.29E-03
104	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	N	1.1E-04	0.07	0.25	0.36	1.72	0.01	0.55	2.67	10.24	3.63E-07	6.82E-06	5.76E-05	1.22E-03	5.21E-10	1.47E-07	4.93E-06	2.36E-04
102	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	N	1.0E-04	0.03	0.11	0.36	1.62	0.00	0.32	3.05		3.63E-07	6.82E-06	5.76E-05	1.22E-03	5.21E-10	1.47E-07	4.93E-06	2.36E-04
107	Main steam line break with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Y	4.3E-07	0.00	0.00	0.01	0.02	0.03	0.12	0.18	0.22	6.16E-06	5.73E-05	3.48E-04	3.95E-03	5.28E-07	1.03E-05	8.51E-05	1.35E-03
105	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Y	1.1E-05	0	0	0.01	0.02	0	0.01	0.04	0.14	8.62E-09	6.03E-07	5.76E-06	1.45E-03	2.70E-11	3.75E-08	8.38E-07	363E-05
106	Main steam line break with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	N	2.2E-06	0	0	0.01	0.04	0	0	0.08	0.21	3.52E-07	6.92E-06	5.83E-05	1.23E-03	2.79E-10	1.33E-07	4.73E-06	2.35E-04
74	Main steam line break with AFW continuing to feed affected generator	None.	N	1.5E-06	0	0	0	0.01	0	0	0	0.02	1.46E-08	7.29E-07	5.94E-06	1.47E-04	4.12E-12	3.20E-08	8.04E-07	3.66E-05

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
81	Main Steam Line Break with AFW continuing to feed affected generator and with HHSI failure initially.	Operator opens ADVs (on intact generators). HHSI is restored after CFTs discharge 50%.	N	2.7E-06	0	0	0	0	0	0	0	0	0.00E+00	3.56E-13	3.54E-09	2.75E-06	0.00E+00	0.00E+00	0.00E+00	1.24E-09
Oconee Unit 1																				
27	MSLB without trip of turbine-driven emergency feedwater.	Operator throttles HPI to maintain 27.8 K [50° F] subcooling margin.	N	2.1E-06	0	0	0	0	0	0	0	0.01	0.00E+00	3.05E-13	2.80E-07	5.11E-06	0.00E+00	0.00E+00	5.16E-09	3.70E-07
99	MSLB with trip of turbine-driven EFW by MSLB Circuitry	HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	N	2.4E-07	0	0	0	0	0	0	0	0	0.00E+00	0.00E+00	1.19E-07	2.08E-06	0.00E+00	0.00E+00	2.06E-09	4.26E-07
100	MSLB with trip of turbine-driven EFW by MSLB Circuitry	Operator throttles HPI 20 minutes after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Y	5.1E-08	0	0	0	0	0	0	0	0	0.00E+00	0.00E+00	7.81E-08	6.06E-06	0.00E+00	0.00E+00	4.08E-08	3.47E-06
101	MSLB without trip of turbine-driven EFW by MSLB Circuitry	Operator throttles HPI to maintain 27.8 K [50° F] subcooling margin (throttling criteria is 27.8 K [50°F] subcooling).	Y	3.9E-07	0	0	0	0	0	0	0	0	0.00E+00	1.51E-09	1.05E-06	7.58E-06	0.00E+00	0.00E+00	1.06E-09	2.02E-07
Palisades																				
54	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	N	4.3E-06	0.44	0.51	0.66	0.69	1.66	1.88	1.62	1.51	5.16E-05	1.37E-04	9.41E-04	4.88E-03	1.79E-05	6.30E-05	6.12E-04	3.52E-03
27	Main steam line break with controller failure resulting in the flow from two AFW pumps into affected steam generator. Break assumed to be inside containment causing containment spray actuation.	Operator starts second AFW pump.	N	3.7E-05	0.13	0.2	0.37	0.41	0.26	0.41	0.72	0.92	1.56E-06	5.92E-06	7.00E-05	4.71E-04	2.97E-07	1.48E-06	3.23E-05	2.98E-04

26	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	Operator isolates AFW to affected SG at 30 minutes after initiation.	N	5.7E-04	0.04	0.11	0.41	0.7	0.02	0.05	0.29	0.98	4.98E-08	2.41E-07	5.20E-06	5.02E-05	2.13E-09	1.55E-08	1.01E-06	2.07E-05
50	Main steam line break with controller failure resulting in the flow from two AFW pumps into affected steam generator. Break assumed to be inside containment causing containment spray actuation.	Operator starts second AFW pump. Operator does not throttle HPI.	Y	5.8E-07	0.01	0.01	0.01	0.01	0.02	0.02	0.03	0.03	4.45E-06	1.45E-05	1.41E-04	8.92E-04	1.24E-06	4.73E-06	7.46E-05	6.03E-04
51	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	Y	7.5E-08	0	0	0.01	0.01	0.02	0.02	0.02	0.02	2.55E-05	7.09E-05	5.26E-04	2.85E-03	9.92E-06	3.51E-05	3.53E-04	2.05E-03
34	Main steam line break concurrent with a single tube failure in SG-A due to MSLB vibration.	Operator isolates AFW to affected SG at 15 minutes after initiation. Operator trips RCPs assuming that they do not trip as a result of the event. Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS	N	1.5E-05	0	0.01	0.03	0.04	0	0.01	0.03	0.06	1.43E-07	7.88E-07	1.38E-05	1.08E-04	8.33E-09	6.97E-08	3.30E-06	4.77E-05
49	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	Operator isolates AFW to affected SG at 30 minutes after initiation. Operator does not throttle HPI.	Y	1.0E-05	0	0	0	0.01	0	0	0	0.01	9.02E-09	5.39E-08	1.83E-06	1.98E-05	5.90E-10	6.17E-09	3.85E-07	9.10E-06
24	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	None	N	2.4E-06	0	0	0	0	0	0	0	0	4.98E-08	2.41E-07	5.20E-06	5.02E-05	1.67E-09	1.41E-08	9.22E-07	1.96E-05
29	Main steam line break with break assumed to be inside containment causing containment spray actuation.	None. Operator does not throttle HPI.	Y	4.2E-08	0	0	0	0	0	0	0	0	2.02E-07	5.18E-07	3.54E-06	2.36E-05	1.94E-07	4.71E-07	2.16E-06	1.38E-05

Table A.6. Transient descriptions and FAVOR 04.1 results for SO-2 transients involving all (or a very large number of) stuck-open valves

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																				
110	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes	Operator controls HHSI 60 minutes after allowed.	N	6.9E-04	0	0	0.02	0.32	0	0	0.02	1.43	1.75E-11	2.48E-08	4.71E-07	3.64E-05	0.00E+00	0.00E+00	4.20E-09	3.65E-06
108	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed.	Y	6.5E-04	0	0.01	0.01	0.2	0	0	0.02	0.64	1.75E-11	2.48E-08	4.71E-07	3.64E-05	0.00E+00	0.00E+00	4.20E-09	3.65E-06
111	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Y	6.8E-05	0	0.01	0.02	0.11	0	0	0.07	0.51	3.08E-09	2.62E-07	3.45E-06	1.40E-04	1.06E-15	5.14E-10	1.43E-07	2.04E-05
109	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Y	6.8E-05	0	0	0.02	0.12	0	0	0.03	0.43	3.08E-09	2.62E-07	3.45E-06	1.40E-04	1.06E-15	5.14E-10	1.43E-07	2.04E-05
112	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	N	1.4E-05	0	0	0	0.01	0	0	0	0.02	1.86E-11	2.13E-08	3.84E-07	3.55E-05	0.00E+00	3.35E-12	4.82E-09	3.76E-06
118	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator	None.	N	9.3E-06	0	0	0	0.01	0	0	0	0.02	3.14E-10	5.80E-08	1.02E-06	5.40E-05	0.00E+00	2.34E-11	2.44E-08	6.94E-06

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
113	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Y	2.7E-06	0	0	0	0	0	0	0	0.01	4.07E-10	2.35E-07	3.06E-06	1.00E-04	7.76E-14	6.47E-10	4.69E-08	1.17E-05
78	Reactor/turbine trip with failure of MFW and AFW.	Operator opens all ASDVs to let condensate fill SGs.	N	3.3E-8	0	0	0	0	0	0	0	0.00	0	0	0	0	0	0	0	1.0E-19
Oconee Unit 1																				
89	Reactor/turbine trip with loss of MFW and EFW.	Operator opens all TBVs to depressurize the secondary side to below the condensate booster pump shutoff head so that these pumps feed the steam generators. Booster pumps are assumed to be initially uncontrolled so that the steam generators are overfilled	N	5.4E-07	0	0	0	0	0	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
98	Reactor/turbine trip with loss of MFW and EFW	Operator opens all TBVs to depressurize the secondary side to below the condensate booster pump shutoff head so that these pumps feed the steam generators. Booster pumps are assumed to be initially uncontrolled so that the steam generators are overfilled	Y	1.0E-07	0	0	0	0	0	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Palisades																				
22	Turbine/reactor trip with loss of MFW and AFW.	Operator depressurizes through ADVs and feeds SGs using condensate booster pumps. Operators maintain a cooldown rate within technical specification limits and throttle condensate flow at 84% level in the steam generator.	N	6.7E-05	0	0	0	0	0	0	0	0.01	0.00E+00	1.38E-12	3.12E-08	1.20E-06	0.00E+00	4.25E-13	5.61E-09	7.07E-07

Table A.7. Transient descriptions and FAVOR 04.1 results for SO-2 transients involving just a few (one or two) stuck-open secondary valves

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																				
No transients of this type were analyzed																				
Oconee Unit 1																				
28	Reactor/turbine trip with 1 stuck-open safety valve in SG-A.	None	N	7.5E-08	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29	Reactor/turbine trip with 1 stuck-open safety valve in SG-A and a second stuck-open safety valve in SG-B.	None	N	3.1E-07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30	Reactor/turbine trip with 1 stuck-open safety valve in SG-A.	None	Y	1.5E-07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
31	Reactor/turbine trip with 1 stuck-open safety valve in SG-A and a second stuck-open safety valve in SG-B.	None	Y	8.4E-09	0	0	0	0	0	0	0	0	0	0	0	6.2E-11	0	0	0	6.2E-11
36	Reactor/turbine trip with 1 stuck-open safety valve in SG-A and a second stuck-open safety valve in SG-B.	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8-cm [120-in.] pressurizer level.	N	1.4E-05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
37	Reactor/turbine trip with 1 stuck-open safety valve in SG-A.	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8-cm [120-in.] pressurizer level.	Y	1.4E-06	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
38	Reactor/turbine trip with 1 stuck-open safety valve in SG-A and a second stuck-open safety valve in SG-B.	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8-cm [120-in.] pressurizer level.	Y	2.7E-06	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
90	Reactor/turbine trip with 2 stuck-open safety valves in SG-A.	Operator throttles HPI 20 minutes after 2.7 K [5°F] subcooling and 254-cm [100"] pressurizer level is reached [throttling criteria is 27.8 K [50°F] subcooling].	N	6.3E-07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
102	Reactor/turbine trip with 2 stuck-open safety valves in SG-A.	Operator throttles HPI 20 minutes after 2.77 K [5°F] subcooling and 254-cm [100-in.] pressurizer level is reached (throttling criteria is 27 K [50°F] subcooling).	Y	2.0E-07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Palisades																				
P55	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump.	N	2.7E-03	0.6	1.7	5.0	5.4	3.0	8.4	16.8	18.0	3.5E-08	3.0E-07	5.9E-06	3.8E-05	1.9E-08	1.9E-07	5.0E-06	3.6E-05
19	Reactor trip with 1 stuck-open ADV on SG-A.	None. Operator does not throttle HPI.	Y	2.3E-03	0.2	0.8	2.5	3.1	0.9	3.2	7.4	9.7	9.1E-09	1.1E-07	2.4E-06	1.6E-05	3.6E-09	5.4E-08	1.9E-06	1.5E-05
52	Reactor trip with 1 stuck-open ADV on SG-A. Failure of both MSIVs (SG-A and SG-B) to close.	Operator does not isolate AFW on affected SG. Normal AFW flow assumed (200 gpm). Operator does not throttle HPI.	Y	6.4E-04	0	0.0	0.1	0.4	0.0	0.1	0.4	1.2	1.7E-08	1.5E-07	3.0E-06	1.9E-05	6.6E-09	7.6E-08	2.3E-06	1.8E-05
16	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump. Operator isolates AFW to affected SG at 30 minutes after initiation. Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to main	N	1.2E-04	0	0	0	0	0	0	0	0	0	9.4E-12	2.3E-08	8.6E-07	0	6.4E-13	2.1E-09	2.7E-07

TH Transient #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
18	Turbine/reactor trip with 1 stuck-open ADV on SG-A. Failure of both MSIVs (SG-A and SG-B) to close.	Operator does not isolate AFW on affected SG. Normal AFW flow assumed (200 gpm). Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level	N	4.7E-03	0	0	0	0	0	0	0	0	0	2.1E-12	3.0E-11	8.3E-09	0	8.8E-13	2.9E-11	3.3E-09

Table A.8. Transient descriptions and FAVOR 04.1 results for feed and bleed, overfeed, and steam generator tube rupture transients

Plant	Class	TH #	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
							32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	1000	32	60	Ext-A	Ext-B
Palisades	F&B	31	Turbine/reactor trip with failure of MFW and AFW. Containment spray actuation assumed due to PORV discharge.	Operator maintains core cooling by “feed and bleed” using HPI to feed and two PORVs to bleed.	N	1.29E-05	0.05	0.05	0.07	0.07	0.06	0.08	0.10	0.09	2.09E-06	5.34E-06	3.82E-05	2.04E-04	2.36E-07	1.08E-06	1.54E-05	9.24E-05
Palisades	F&B	32	Turbine/reactor trip with failure of MFW and AFW. Containment spray actuation assumed due to PORV discharge.	Operator maintains core cooling by “feed and bleed” using HPI to feed and two PORV to bleed. AFW is recovered 15 minutes after initiation of “feed and bleed” cooling. Operator closes PORVs when SG level reaches 60%.	N	1.08E-06	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	3.87E-09	6.77E-08	2.26E-06	1.70E-05	1.50E-09	3.30E-08	1.71E-06	1.58E-05
Beaver	F&B	31	Reactor/turbine trip w/ feed & bleed	Operator opens all pzr PORVs & uses all charging/HHSI pumps	N	3.10E-7	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	3.44E-7	3.28E-6	1.76E-5	1.72E-4	3.10E-12	4.02E-9	3.32E-8	1.39E-6
Beaver	Overfeed	76	Reactor/turbine trip w/full MFW to all 3 SGs (MFW maintains SG level near top).	Operator trips reactor coolant pumps.	Y	1.1E-04	0	0	0.00	0.00	0	0	0.00	0.00	0	0	1.14E-10	1.74E-6	0	0	8.92E-15	9.73E-08
Oconee	SGTR	127	SGTR with a stuck-open SRV in SG-B. A reactor trip is assumed to occur at the time of the tube rupture. Stuck safety relief valve is assumed to reclose 10 minutes after initiation.	Operator trips RCPs 1 minute after initiation. Operator also throttles HPI 10 minutes after 2.77 K [5° F] subcooling and 254-cm [100-in.] pressurizer level is reached (assumed throttling criteria is 27 K [50°F] subcooling).	Y	1.25E-07	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table A.9. Transient descriptions and FAVOR 04.1 results for mixed primary and secondary initiator transients

#	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Beaver Valley Unit 1																				
B-65	Reactor/turbine trip w/two stuck-open pressurizer SRVs and HHSI failure	Operator opens all ASDVs 5 minutes after HHSI would have come on.	N	1.04E-09	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	1.71E-09	1.43E-07	1.19E-06	3.64E-05	0.00E+00	0.00E+00	6.42E-14	2.98E-08
68	Reactor/turbine trip w/two stuck-open pressurizer SRVs that reclose at 6000 s with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have come on.	N	1.33E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	1.41E-07	5.69E-07	2.36E-06	5.94E-05	1.36E-07	2.69E-07	1.16E-07	1.51E-06
72	Reactor/turbine trip w/one stuck-open pressurizer SRV with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have come on.	N	5.14E-07	-	-	-	-	-	-	-	-	-	-	-	1.19E-12	-	-	-	-
73	Reactor/turbine trip w/one stuck-open pressurizer SRV with HHSI failure	Operator open all ASDVs 5 minutes after HHSI would have come on.	Y	6.55E-08	-	-	-	-	-	-	-	-	-	-	-	4.64E-08	-	-	-	-
82	Reactor/turbine trip w/one stuck-open pressurizer SRV (recloses at 6000 s) and with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have started.	N	1.51E-06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	2.03E-11	0.00E+00	2.40E-09	0.00E+00	1.92E-11	0.00E+00	2.30E-09
83	2.54-cm [1.0-in.] surge line break with HHSI failure and motor driven AFW failure. MFW is tripped. Level control failure causes all steam generators to be overfed with turbine AFW, with the level maintained at top of SGs.	Operator trips RCPs. Operator opens all ASDVs 5 minutes after HHSI would have come on.	N	3.51E-06	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

#	System Failure	Operator Action	HZP	IEF	Percent Contribution to Total Frequency of Crack Initiation (FCI)				Percent Contribution to Total Through Wall Cracking Frequency (TWCF)				Mean CPI				Mean CPTWC			
					32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B	32	60	Ext-A	Ext-B
Oconee Unit 1																				
O-110	5.08-cm [2 in.] surge line break with HPI failure	At 15 minutes after transient initiation, operator opens both TBV to lower primary system pressure and allow CFT and LPI injection.	N	3.42E-06	0.18	0.46	1.18	1.48	0.00	0.00	0.64	1.16	7.19E-08	1.02E-06	2.62E-04	1.73E-03	4.93E-13	6.88E-10	2.16E-06	3.11E-05
120	2.54-cm [1-in.] surge line break with HPI Failure	At 15 minutes after sequence initiation, operators open all TBVs to depressurize the system to the CFT setpoint. When the CFTs are 50% discharged, HPI is assumed to be recovered. The TBVs are assumed remain opened for the duration of the transient	Y	4.22E-08	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01	1.69E-12	7.80E-09	2.69E-06	2.50E-05	1.31E-12	7.03E-09	2.51E-06	2.28E-05
44	2.54-cm [1-in.] surge line break with HPI Failure	At 15 minutes after initiation, operators open all TBVs to depressurize the system to the CFT setpoint. When the CFTs are 50% discharged, HPI is assumed to be recovered. The TBVs are assumed remain open for the duration of the transient.	N	2.69E-07	0.00	0.00	0.00	0.00	0.00	-	-	0.01	0.00E+00	0.00E+00	4.41E-07	6.48E-06	-	-	4.27E-07	6.18E-06
119	2.54-cm [1-in.] surge line break with HPI Failure	At 15 minutes after transient initiation, the operator opens all turbine bypass valves to lower primary system pressure and allow core flood tank and LPI injection.	Y	4.41E-07	0.00	0.00	0.00	0.01	0.00	-	0.00	0.01	2.83E-10	1.36E-08	8.48E-06	5.52E-05	-	4.42E-14	1.81E-07	2.95E-06
8	2.54-cm [1-in.] surge line break with 1 stuck-open safety valve in SG-A.	None	N	9.68E-08	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
12	2.54-cm [1-in.] surge line break with 1 stuck-open safety valve in SG-A.	HPI throttled to maintain 27.8 K [50° F] subcooling margin	N	9.24E-07	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

15	2.54-cm [1-in.] surge line break with HPI Failure	At 15 minutes after transient initiation, operator opens all TBVs to lower primary system pressure and allow CFT and LPI injection.	N	3.39E-08	-	-	0.00	0.00	-	-	-	0.00	-	-	1.74E-09	5.79E-07	-	-	-	1.33E-09
111	2.54-cm [1-in.] surge line break with HPI failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. At 3000 seconds after initiation, operator starts throttling HPI to 55 K [100°F] subcooling	N	4.16E-07	-	-	0.00	0.00	-	-	-	0.00	-	-	1.01E-10	1.83E-07	-	-	-	2.07E-09
117	Stuck-open pressurizer safety valve and HPI failure	At 15 minutes after initiation, operator opens all TBV to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. The SRV is closed 5 minutes after HPI recovered. HPI is throttled at 1 minute after 2.	N	5.38E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.01	2.12E-11	7.19E-09	4.21E-06	7.37E-05	0.00E+00	0.00E+00	8.73E-09	7.36E-07
116	Stuck-open pressurizer safety valve and HPI failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. The HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254-cm [100”] pressure	N	2.60E-07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	0.00E+00	1.40E-07	6.43E-06	0.00E+00	0.00E+00	1.15E-10	6.68E-08
125	Stuck-open pressurizer safety valve and HPI Failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254-cm [100”] pressurize	Y	4.61E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	1.44E-10	4.89E-09	6.24E-06	5.54E-05	0.00E+00	0.00E+00	7.42E-08	2.34E-06
126	Stuck-open pressurizer safety valve and HPI Failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. SRV is closed at 5 minutes after HPI is recovered. HPI is throttled at 1 minute after	Y	8.41E-08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.00	0.00E+00	0.00E+00	5.26E-08	2.31E-06	0.00E+00	0.00E+00	7.67E-11	8.95E-08